ANL-7597

Argonne National Laboratory

SURVEILLANCE OF EBR-II BLANKET SUBASSEMBLIES

by

V. G. Eschen

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Printed in the United States of America
Available from
Clearinghouse for Federal Scientific and Technical Information
National Bureau of Standards, U. S. Department of Commerce
Springfield, Virginia 22151
Price: Printed Copy $3.00; Microfiche $0.65

ANL-7597
Reactor Technology

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

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EBR-II Project

September 1969

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ABSTRACT

This report covers examinations performed on unalloyed, depleted-uranium blanket elements from Experimental Breeder Reactor-II (EBR-II). The criteria used in arriving at the present burnup limit of 0.27 at. % (max) are discussed and shown to be conservatively established. Swelling rates were investigated by calculating R values (ratios of swelling percentage to burnup percentage); the minimum and maximum values of R calculated for the points of highest burnup were 63 and 148. A maximum cladding strain of 0.9% was indicated at 0.27 at. % (max) burnup.

Specific recommendations are presented for more fully evaluating the irradiation behavior and improving the performance of the blanket elements. A program to evaluate the feasibility of increasing the burnup limit of the present blanket material is proposed.

I. INTRODUCTION

Experimental Breeder Reactor-II (EBR-II) is a sodium-cooled reactor whose original purpose was to demonstrate the feasibility of using a breeder reactor in a power station. This feasibility has now been amply demonstrated, and EBR-II, while continuing to produce electrical power, is serving primarily as an irradiation facility for breeder-reactor fuels and materials.

In common with other breeder-reactor designs, EBR-II has its core surrounded by a thick blanket of depleted uranium. The axial blanket material has been removed from EBR-II for convenience in the irradiation program, but most of the radial-blanket material remains as originally designed. Extensive experience has been accumulated with this radial blanket; EBR-II began power operation in 1964, and some of the blanket subassemblies removed for examination have been exposed to reactor operation of 15,000 MWd and more.

In future commercial breeders, the reprocessing of the blanket to reclaim plutonium will be an important part of the economics of the fuel cycle. The frequency of reprocessing will depend partly on the metallurgical performance of the blanket subassemblies, i.e., on their stability and integrity under extended irradiation. This report deals with data from EBR-II that can be relevant to the operation of future breeder reactors, as well as to the more limited area of improving the blanket performance in EBR-II.

II. DESCRIPTION OF CORE AND BLANKET

Subassemblies are arranged in a hexagonal array to make up the EBR-II core. There are 15 rows or rings of subassemblies in the reactor. In the first core loading, Rows 1-5 contained the "active" or driver-fuel subassemblies, Row 6 was a mixture of driver-fuel and blanket subassemblies, and Rows 7-15 were entirely blanket subassemblies. The active core height was 14.25 in. Rows 6 and 7 are known as the inner-blanket region, and Rows 8-15 are the outer-blanket region. Reference 1 contains a more complete description of the core configuration.

Each blanket subassembly contains 19 elements, and each element contains five depleted-uranium slugs composed of 0.2% ^{235}U, and the balance ^{238}U. Each slug is 0.433 in. in diameter and 11 in. long. Sodium provides the heat-transfer bond between the uranium and the Type 304 stainless steel cladding. Figure 1 shows the design of the blanket subassembly and illustrates the difference in inner- and outer-blanket subassemblies, namely, the difference in the design of the lower adapter of the subassembly. The orificing and coolant-flow characteristics of the lower adapters are considerably different.

The reactor was operated for approximately 15,000 MWd with the core essentially the same as the original design. Examination of a Row 6 blanket subassembly (A-701) after this period of time showed that the uranium slugs had swelled considerably. To evaluate the condition of the inner- and outer-blanket subassemblies, more examinations were performed. This report presents the results of these examinations.

III. PURPOSES OF THE INVESTIGATION

The purposes of this investigation were to:

1. Determine what, if any, physical changes had occurred in the blanket elements and in the uranium slugs as shown by diameter measurements, density determinations, metallography, and electron microscopy, and to correlate these measurements with burnup of the uranium.

2. Evaluate how much, if any, cladding strain had occurred and correlate it with burnup.

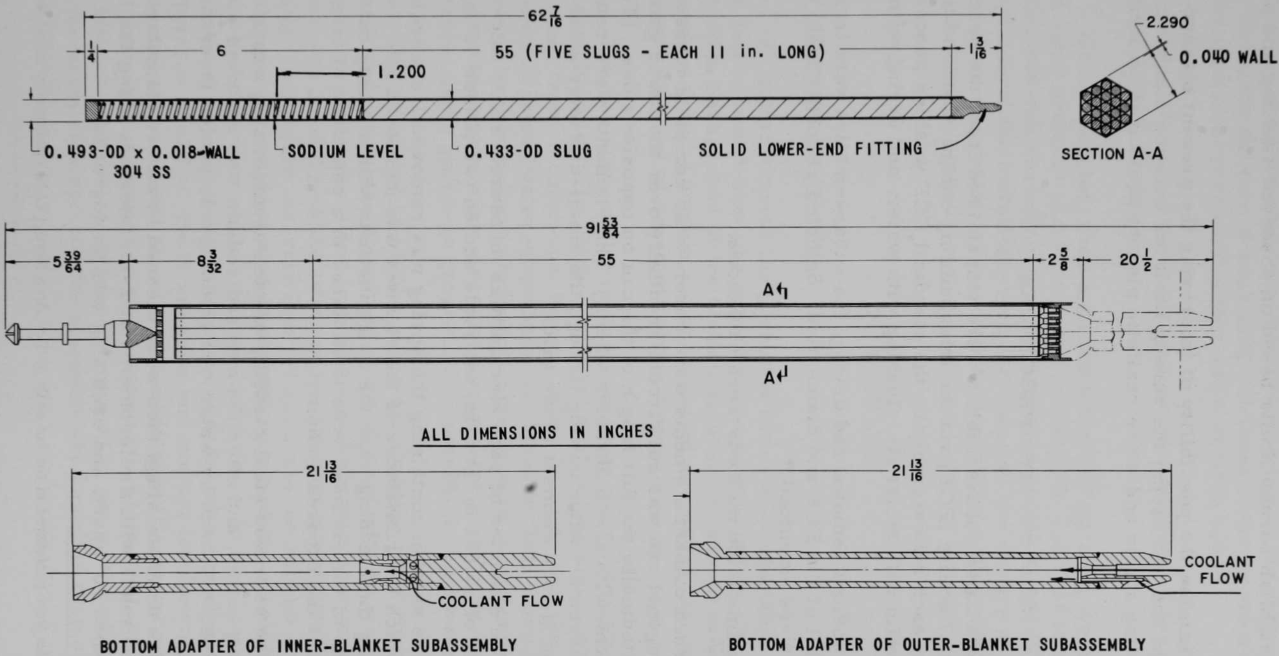

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Fig. 1. EBR-II Inner- and Outer-blanket Subassemblies[1]

3. Establish burnup limits based on observed swelling and plenum pressure.

4. Examine the possibility of improving the present blanket-element design and increasing the burnup limit for existing blanket material by using cladding strain and more realistic plenum pressures as limiting criteria.

IV. PROCEDURE

The irradiated subassemblies were transferred from the reactor to the Fuel Cycle Facility (FCF) via the interbuilding coffin. Sodium was removed from each subassembly by the standard FCF washing procedure: purging the coffin with moist air, flushing with water, and drying with air.

After sodium removal and drying, the subassemblies were transferred into the air cell of the FCF and dismantled. Selected elements from each subassembly were examined.*

The examination methods were as follows:

1. When blanket elements were received as complete subassemblies, the outer hexagonal can was cut circumferentially over the lower grid and then cut longitudinally the full length of the can on opposite sides. (This cutting was done with a thin abrasive wheel.) The elements were removed from the subassembly after cutting through the small-diameter solid end fittings attaching each element to the grid.

2. The diameter of each element was measured with a micrometer to an accuracy of ±0.001 in. at 6-in. intervals and on two planes 90° apart.

3. The section containing the spring was removed from each element, after which the remainder of the element was sectioned into five sections by cutting the cladding near the slug junctions with a tubing cutter. These sections of the element were then placed in a container of isoamyl alcohol to react the exposed sodium.

4. The stainless steel cladding of each uranium slug was slit with a small milling cutter, and after the exposed sodium was removed by rinsing in alcohol followed by water, the cladding was peeled from the slug.

5. The uranium slugs then were cleaned in water, diameter measurements were made with a micrometer at 2-in. intervals along the length and in two planes 90° apart, and each slug was photographed.

---

*These examinations were performed in the hot cells of Test Area North (TAN) operated by Idaho Nuclear Corporation.

6. After the slugs were bathed in acetone to remove any moisture from them, the density of each slug was determined by weighing it first in air and then in a mixture of water and a wetting agent.

7. After the density measurements, samples of the slugs were taken for burnup analysis.

Other samples, each approximately 1/2 in. long, were cut from the 11-in.-long uranium slugs, mounted in Bakelite, and sent to Argonne, Illinois, for electron-microscopy and metallographic examinations.* The polishing and etching of the metallographic samples and the replica preparation for the electron microscopy were done as described in the memorandum reprinted in the appendix.

V. RESULTS

A. Fabrication Information

To better discuss the problems involved in analyzing the postirradiation data obtained from the examinations of the blanket elements, the fabrication background of the blanket elements is reviewed briefly here. References 2 and 3 contain a more detailed account of fabrication of the uranium slugs and of assembly of the elements.

The depleted unalloyed-uranium slugs were fabricated and tested at Argonne, Illinois. The slugs consist of depleted-uranium cylinders formed by rolling from vacuum-cast ingots. After being rolled, the slugs were beta-heat treated (water-quenched), machined to final dimensions, and inspected by ultrasonic techniques for grain-size variations and casting defects. The slugs were also spot-checked for length and diameter conformance, using go/no-go gauges. The individual slugs were not measured.

The stainless steel tubes used for cladding were cut to length and cleaned, and the inside diameters were checked with an air gauge. The bottom end plug was welded into place, and the weld was leak-checked. Sodium was loaded into the tubes under an inert-gas atmosphere. The uranium slugs were inserted into the tube, and the sodium was melted. The compression spring was positioned on top of the uranium slugs, and the top end plug was welded on. After the top weld was leak-tested, the elements were bonded by heating to 475°C for 4 hr and vibrating (at temperature) at 1800 Hz. Bond integrity and sodium level were determined by passing the elements through an encircling differential eddy-current coil. Individual identities of the uranium slugs were not maintained during the element-fabrication steps. Consequently, the preirradiation data for both the uranium slugs and the elements are based on manufacturing specifications.

*This work was performed by W. F. Murphy of the Metallurgy Division at Argonne, Illinois.

Table I contains the analytical data and specified compositions for the ingot material. Of interest is the fact that carbon additions were deliberately made to maintain carbon contents in the range from 200 to 500 ppm.[2] Table II lists the dimensional specifications used in fabricating the uranium slugs and the Type 304 stainless steel cladding.

TABLE I. Analytical Results and Specified Compositions for Depleted-uranium Ingots[2]

Element	Specification, ppm max unless Otherwise Identified	Ingot Analyses,[a] ppm unless Otherwise Identified		
		High	Low	Average
^{235}U	0.22 ± 0.02 wt %	0.2312 wt %	0.2087 wt %	0.2148 wt %
Boron	1	1	NF	NMA
Cadmium	1	1	NF	NMA
Carbon	750 (preferably 200-500)	740	60	308
Chromium	100	20	NF	NMA
Copper	100	70	4	15
Iron + nickel	300	290	70	160
Magnesium	25	10	5	NMA
Manganese	150	140	2	48
Nitrogen	100	88	10	24
Silicon	150	160[b]	15	75
All others[c]	400	280	7	82

[a]Based on 225 ingots.
[b]One ingot analyzed 160 ppm, all others below 150 ppm.
[c]Includes aluminum, beryllium, cobalt, lead, phosphorous, tin, vanadium and zinc.
NF Not found.
NMA No meaningful average because analyses were usually given as "less than... ."

TABLE II. Dimensional Specifications for Depleted-uranium Blanket Elements and Slugs[2]

Cladding OD,[a] in.	0.493 ± 0.001
Cladding ID,[a] in.	0.457 ± 0.001
Slug OD, in.	0.433 ± 0.001
Slug length, in.	11.0 (nominal)
Slug density, g/cm^3	
Batch average	18.96
Range	18.89-19.03

[a]Obtained from ANL Drawing EB-1-25061-D.

B. Postirradiation Examination of Blanket Elements and Uranium Slugs

Thirty elements selected from nine subassemblies were examined. Dimensional measurements of the OD of the 150 uranium slugs from these elements revealed that only three subassemblies contained uranium slugs with measurable diameter increases. Table III gives the results from the slugs that showed diameter increases. Figures 2 and 3 illustrate the relationship observed between the diameter measurements and the vertical position along the slugs.

TABLE III. Summary of Postirradiation Data Obtained from EBR-II Depleted-uranium Blanket Elements

Subassembly No.	Reactor Grid Location	Element No.	Cladding OD, in. Inches from Top of Element										Slug[a] No.	Slug Diameter, in. Inches from Bottom of Element						Slug Length, in.	Slug Density,[b] g/cm³	Calculated Avg Burnup, at. %
			6	12	18	24	30	36	42	48	54	60		0	2	4	6	8	10			
A-701	6A5	1	No measurements taken; cladding damaged										2	0.446 0.446	0.442 0.438	0.438 0.438	0.435 0.433	0.434 0.434	0.432 0.433	11.140	18.3087	0.037
													3	0.442 0.445	0.441 0.447	0.443 0.439	0.454 0.440	0.455 0.448	0.454 0.438	11.040	18.1330	0.118
													4	0.433 0.433	0.433 0.433	0.433 0.433	0.435 0.434	0.435 0.435	0.438 0.439	10.991	18.8943	0.017
		19	0.491 0.490	0.491 0.491	0.492 0.493	0.493 0.493	0.493 0.495	0.495 0.496	0.493 0.493	0.492 0.492	0.4915 0.492	0.492 0.4915	2	0.4535 0.4515	0.4515 0.451	0.433 0.442	0.437 0.440	0.434 0.4325	0.433 0.433	11.158	18.0334	0.073
													3	0.4515 0.451	0.460 0.4585	0.4625 0.461	0.461 0.461	0.459 0.461	0.458 0.460	11.060	17.2575	0.226[c] . 211
													4	0.432 0.433	0.432 0.433	0.432 0.434	0.434 0.434	0.4345 0.434	0.438 0.433	11.017	18.8353	0.034
		12	0.495 0.495	0.495 0.494	0.495 0.494	0.495 0.494	0.498 0.494	0.495 0.494	0.494 0.494	0.494 0.494	0.493 0.493	0.494 0.493	2	0.458 0.458	0.455 0.442	0.444 0.4435	0.439 0.436	0.435 0.434	0.433 0.432	11.165	17.9992	0.068
													3	0.439 0.443	0.453 0.459	0.458 0.461	0.460 0.461	0.4555 0.461	0.4555 0.461	11.212	17.0417	0.217[c]
													4	0.432 0.432	0.4325 0.4325	0.433 0.433	0.434 0.434	0.435 0.435	0.443 0.442	10.996	19.0006	0.032
		16	0.492 0.4925	0.492 0.492	0.492 0.492	0.4925 0.493	0.497 0.4925	0.494 0.4935	0.492 0.4915	0.4915 0.4915	0.4915 0.4925	0.4915 0.4915	2	0.4575 0.454	0.449 0.449	0.441 0.437	0.437 0.4345	0.433 0.432	0.431 0.4315	11.138	(1)16.7856 (2)17.7226 (3)18.4783 (4)18.9332	(1)0.153 (2)0.096 (3)0.052 (4)0.031
													3	0.445 0.440	0.457 0.440	0.4605 0.455	0.4585 0.455	0.460 0.455	0.456 0.455	11.172	(1)18.0630 (2)17.3940 (3)16.8919 (4)16.9390	(1)0.164 (2)0.211 (3)0.224 (4)0.208
													4	0.433 0.431	0.432 0.432	0.433 0.432	0.434 0.433	0.439 0.436	0.445 0.440	11.000	(1)18.9159 (2)18.8054 (3)18.8554 (4)18.6945	(1)0.015 (2)0.025 (3)0.045 (4)0.085
		18	0.492 0.492	0.492 0.492	0.491 0.492	0.493 0.4925	0.492 0.4935	0.492 0.492	0.492 0.4915	0.492 0.4915	0.4915 0.492	0.4915 0.491	2	0.455 0.4565	0.4565 0.456	0.454 0.454	0.445 0.445	0.439 0.438	0.435 0.435	11.288	(1)16.5270 (2)16.9448 (3)17.8380 (4)18.4608	(1)0.139 (2)0.087 (3)0.047 (4)0.028
													3	0.441 0.441	0.452 0.445	0.455 0.451	0.455 0.453	0.4555 0.455	0.455 0.455	10.998	(1)18.3364 (2)17.5961 (3)17.2062 (4)17.0651	(1)0.148 (2)0.191 (3)0.203 (4)0.189
													4	0.4315 0.431	0.432 0.4315	0.433 0.432	0.434 0.433	0.434 0.433	0.439 0.4375	10.983	(1)18.8484 (2)18.8910 (3)18.8651 (4)18.6463	(1)0.013 (2)0.022 (3)0.041 (4)0.077
A-752	7A3	4	0.491 0.492	0.4915 0.492	0.492 0.493	0.4935 0.493	0.4935 0.493	0.4935 0.493	0.493 0.493	0.493 0.493	0.493 0.493	0.493 0.493	2	0.435 0.438	0.4325 0.438	0.433 0.435	0.4325 0.433	0.4325 0.4325	0.4325 0.433	11.054	18.7028	0.032
													3	0.435 0.433	0.4365 0.433	0.4365 0.435	0.439 0.436	0.439 0.438	0.437 0.4365	11.009	18.6549	0.119
													4	0.433 0.432	0.432 0.433	0.4325 0.432	0.4325 0.432	0.433 0.432	0.433 0.434	11.000	18.8317	0.021
		12	0.4905 0.491	0.491 0.4905	0.492 0.492	0.492 0.493	0.493 0.493	0.4935 0.493	0.493 0.4915	0.492 0.492	0.492 0.492	0.492 0.492	2	0.4355 0.4345	0.433 0.4335	0.433 0.4335	0.4325 0.4325	0.432 0.433	0.4325 0.432	11.045	18.7838	0.020
													3	0.4335 0.4315	0.434 0.4325	0.4335 0.432	0.435 0.432	0.4345 0.432	0.434 0.432	10.985	18.7671	0.075
													4	0.432 0.4325	0.433 0.434	0.4325 0.4325	0.433 0.434	0.432 0.4335	0.432 0.434	10.985	18.9000	0.013
		13	0.490 0.490	0.491 0.491	0.491 0.492	0.492 0.491	0.492 0.492	0.493 0.4925	0.4925 0.492	0.493 0.493	0.492 0.493	0.492 0.493	2	0.4425 0.444	0.439 0.4405	0.437 0.4365	0.434 0.4345	0.433 0.433	0.4325 0.4325	11.023	18.6321	0.022
													3	0.438 0.4385	0.4375 0.439	0.437 0.436	0.438 0.4375	0.438 0.438	0.436 0.4375	11.064	18.5363	0.083

TABLE III. (Contd.)

Subassembly No.	Reactor Grid Location	Element No.	Cladding OD, in. Inches from Top of Element										Slug[a] No.	Slug Diameter, in. Inches from Bottom of Element						Slug Length, in.	Slug Density,[b] g/cm3	Calculated Avg Burnup, at. %
			6	12	18	24	30	36	42	48	54	60		0	2	4	6	8	10			
A-752 (Contd.)	7A3	13	0.490 0.490	0.491 0.491	0.491 0.492	0.492 0.491	0.492 0.492	0.493 0.4925	0.4925 0.492	0.493 0.493	0.492 0.493	0.492 0.493	4	0.433 0.4325	0.432 0.432	0.432 0.432	0.433 0.4315	0.4325 0.432	0.4335 0.432	10.998	18.9062	0.015
		8	0.492 0.490	0.4915 0.491	0.491 0.491	0.491 0.4915	0.494 0.4925	0.4925 0.494	0.492 0.493	0.491 0.493	0.493 0.493	0.4925 0.493	2	0.455 0.455	0.455 0.446	0.440 0.439	0.436 0.436	0.434 0.434	0.433 0.4325	11.110	(1)17.2903 (2)18.1519 (3)18.7724	(1)0.060 (2)0.028 (3)0.014
													3	0.444 0.441	0.448 0.451	0.454 0.453	0.442 0.455	0.448 0.457	0.451 0.451	11.130	(1)18.5499 (2)17.9970 (3)17.5903 (4)17.4373	(1)0.082 (2)0.100 (3)0.104 (4)0.095
													4	0.434 0.429	0.434 0.436	0.434 0.436	0.436 0.434	0.437 0.438	0.440 0.445	11.025	18.7836	0.018
A-760	7F4	3	0.4925 0.494	0.493 0.493	0.493 0.493	0.493 0.4925	0.4925 0.4925	0.4925 0.492	0.4935 0.4925	0.493 0.4925	0.493 0.4925	0.492 0.4925	2	0.4375 0.4375	0.436 0.4365	0.4355 0.435	0.4355 0.4335	0.4325 0.433	0.4325 0.432	11.083	18.7104	0.021
													3	0.4335 0.433	0.4375 0.4335	0.437 0.4365	0.438 0.437	0.4395 0.4395	0.438 0.4395	11.024	18.5166	0.079[c]
													4	0.432 0.4325	0.4325 0.432	0.4325 0.433	0.4325 0.433	0.4335 0.4325	0.4315 0.4335	10.998	18.9198	0.014
		6	0.4915 0.492	0.492 0.492	0.492 0.492	0.4925 0.492	0.494 0.495	0.494 0.493	0.492 0.4925	0.493 0.4925	0.493 0.492	0.492 0.493	2	0.440 0.437	0.437 0.436	0.434 0.4335	0.432 0.432	0.432 0.432	0.4315 0.432	11.031	18.6805	0.029
													3	0.436 0.438	0.441 0.446	0.444 0.450	0.455 0.4525	0.449 0.451	0.449 0.4495	11.033	17.7002	0.109
													4	0.432 0.432	0.432 0.4325	0.432 0.4325	0.433 0.432	0.4335 0.433	0.435 0.435	10.993	18.8827	0.019
		17	0.4935 0.494	0.4945 0.494	0.494 0.4935	0.4935 0.4945	0.4955 0.494	0.495 0.4955	0.495 0.496	0.495 0.495	0.4955 0.495	0.4935 0.4935	2	0.448 0.453	0.4465 0.4475	0.440 0.4375	0.4355 0.438	0.4335 0.4345	0.4325 0.4325	11.118	18.7104	0.053
													3	0.443 0.443	0.452 0.4525	0.4525 0.454	0.454 0.455	0.4545 0.4555	0.4546 0.457	11.150	18.5166	0.200[c]
													4	0.4325 0.4325	0.432 0.432	0.433 0.4325	0.4345 0.433	0.4365 0.436	0.439 0.439	10.998	18.9198	0.035
		19	0.493 0.494	0.494 0.493	0.494 0.494	0.494 0.494	0.495 0.496	0.496 0.496	0.494 0.494	0.493 0.493	0.493 0.492	0.493 0.493	2	0.4515 0.4465	0.4425 0.442	0.4365 0.436	0.434 0.434	0.433 0.4325	0.4325 0.4325	11.107	18.3239	0.042
													3	0.444 0.4425	0.450 0.449	0.456 0.453	0.457 0.454	0.456 0.454	0.453 0.449	10.980	17.8585	0.159
													4	0.432 0.432	0.433 0.433	0.433 0.433	0.433 0.433	0.435 0.4345	0.438 0.4375	11.000	18.8779	0.028
		13	0.493 0.492	0.493 0.493	0.493 0.493	0.4935 0.4935	0.495 0.494	0.493 0.494	0.4925 0.493	0.493 0.4925	0.493 0.4925	0.493 0.4925	2	0.444 0.445	0.4385 0.439	0.4345 0.434	0.433 0.432	0.432 0.431	0.430 0.431	11.066	(1)17.6268 (2)18.3833 (3)18.8201 (4)18.937	(1)0.111 (2)0.062 (3)0.037 (4)0.023
													3	0.445 0.440	0.455 0.454	0.455 0.454	0.4575 0.4565	0.456 0.4565	0.454 0.4575	11.206	(1)18.0998 (2)17.0261 (3)16.7049 (4)16.4498	(1)0.140 (2)0.171 (3)0.179 (4)0.161
													4	0.4325 0.431	0.431 0.430	0.432 0.431	0.432 0.430	0.435 0.433	0.440 0.440	11.040	(1)19.2470 (2)18.9954 (3)19.1722 (4)18.8306	(1)0.014 (2)0.023 (3)0.046 (4)0.093
		18	0.493 0.494	0.493 0.4925	0.493 0.492	0.4925 0.4925	0.493 0.4915	0.4915 0.4925	0.492 0.492	0.492 0.4915	0.4915 0.492	0.492 0.4925	2	0.459 0.4565	0.4545 0.451	0.443 0.440	0.437 0.436	0.434 0.434	0.433 0.433	11.141	(1)16.5759 (2)18.2738 (3)18.8949 (4)18.9116	(1)0.111 (2)0.062 (3)0.037 (4)0.023
													3	0.4335 0.446	0.422 0.454	0.451 0.458	0.453 0.458	0.454 0.458	0.453 0.4525	11.096	(1)18.3440 (2)17.5191 (3)17.1172 (4)16.9477	(1)0.140 (2)0.171 (3)0.179 (4)0.161
													4	0.4315 0.431	0.432 0.432	0.4325 0.432	0.433 0.431	0.4345 0.435	0.4375 0.4375	11.014	(1)19.2412 (2)18.9848 (3)19.0252 (4)18.7884	(1)0.014 (2)0.023 (3)0.046 (4)0.093

[a] Slug No. 2 is the second one from the top, and slug No. 4 is the second one from the bottom of the element.
[b] No. 1 is the lower end of the slug, and No. 4 the upper end of the slug.
[c] Determined from analytical data

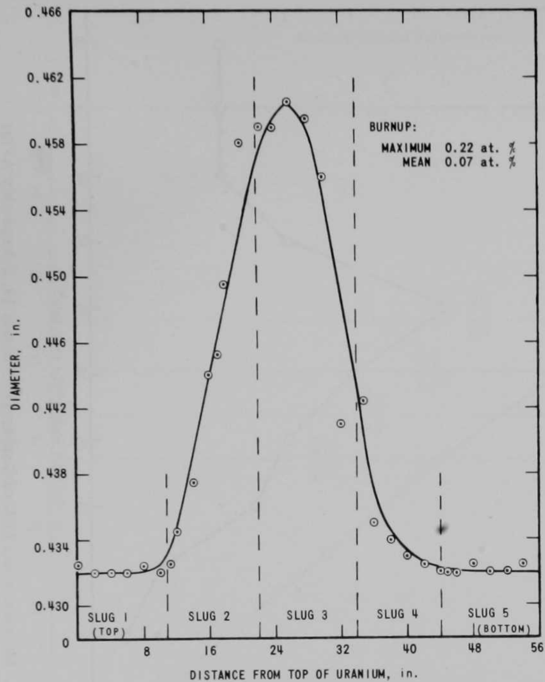

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Fig. 2. Diameter Measurements of Uranium Slugs from Element 12, Subassembly A-701

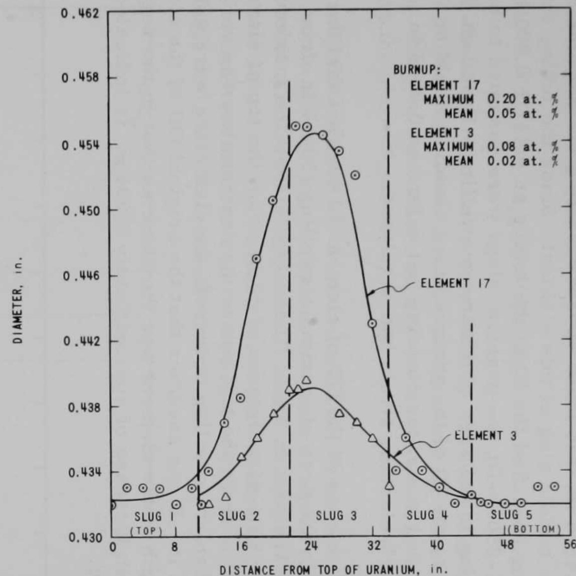

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Fig. 3. Diameter Measurements of Uranium Slugs from Elements 17 and 3, Subassembly A-760

The maximum slug diameter was found in element 19 from subassembly A-701, which had been in position 6A5 in the reactor. A maximum diameter of 0.4625 in. was found approximately 4 in. from the bottom of slug No. 3, the middle slug of this element. Since the tubing specification for the cladding specified the ID of the tubing as 0.457 ± 0.001 in. (ANL Drawing EB-1-25061-D), the uranium slugs were assumed to be in contact with the cladding in the area of maximum swelling. Based on the maximum ID specification, the ID of the preirradiated cladding would be 0.458 in. Therefore, on that basis, the cladding had deformed 0.0045 in. during irradiation, and 0.9% cladding strain had developed.

Measurements of the OD of element 19 verified that the cladding had deformed in the area of maximum swelling (30-40 in. from the top of the element). The data in Table III and Fig. 4 show the maximum cladding OD* to be 0.4955 in. at a distance of 36 in. from the top of element 19. At 6 in. and at 60 in. from the top, where the neutron flux was very low and no measurable uranium swelling occurred, the diameters were approximately 0.491 to 0.492 in. If one assumes that the original OD of the cladding over the entire length of the element was the same as that in the regions of low neutron flux, an increase of approximately 0.004 in. is indicated in the OD of the cladding.

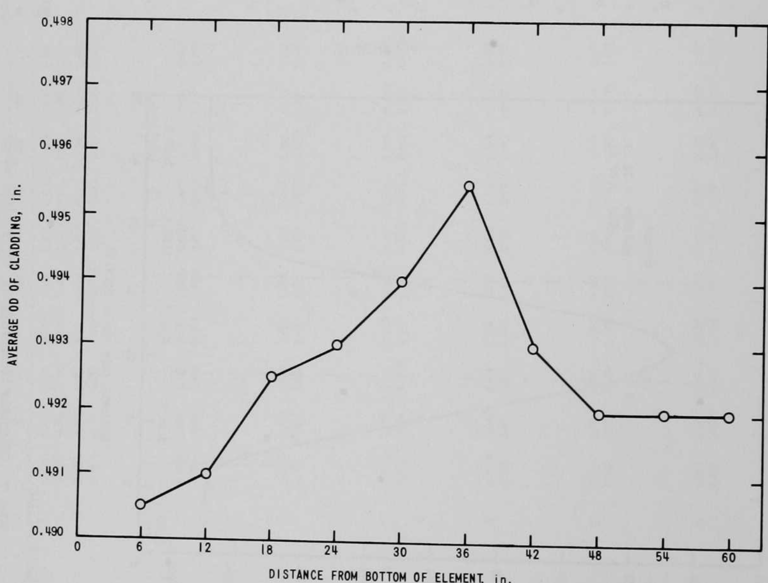


Fig. 4. Measurements of OD of Cladding of Element 19, Subassembly A-701

*Based on the average of two measurements 90° apart.

Using the average density of 18.96 g/cm^2 given in Ref. 2 for the uranium slugs, the percentages of volume change for the irradiated slugs were calculated from the relationship

$$\% \Delta V/V = [(18.96 - d_2)/18.96][100],$$

where d_2 is the density of an irradiated slug. In some cases, the 11-in. slugs were cut into sections about $2\frac{1}{2}$ in. long and the density measured. The volume changes were calculated in the same manner as for complete slugs, using the above formula.

Burnup determinations were made either from analytical determinations based on the plutonium content in a sample or by calculations using radial and axial fission distributions as a function of element and slug position in the reactor. The radial fission distribution is given in Fig. 5, and the axial fission rates for selected positions in Rows 6, 7, and 9 are shown in Figs. 6, 7, and 8, respectively.

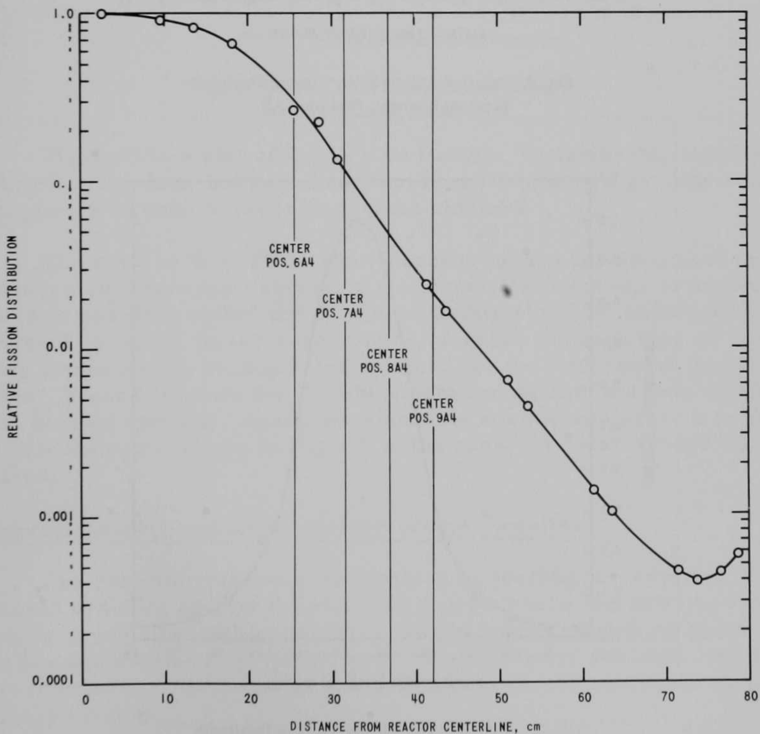


Fig. 5. Radial Fission Distribution in EBR-II Core and Blanket

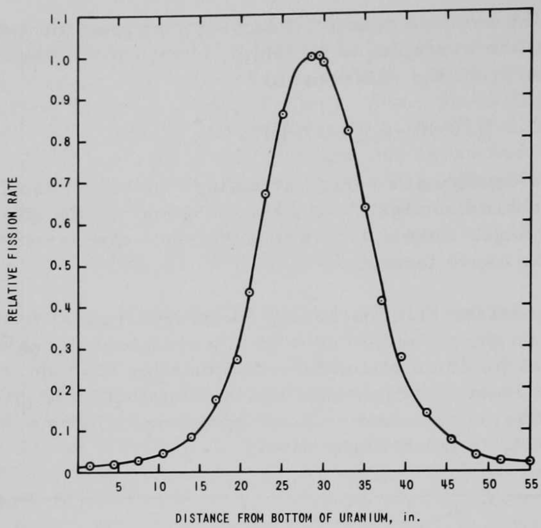


Fig. 6. Relative Axial Fission Rates in Depleted-uranium Element, Position 6A3

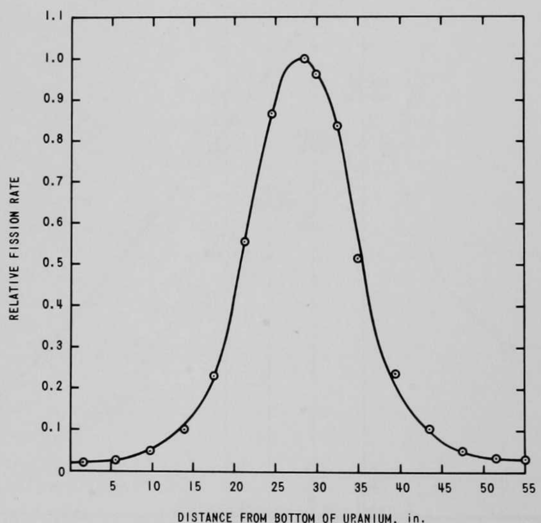


Fig. 7. Relative Axial Fission Rates in Depleted-uranium Element, Position 7A4

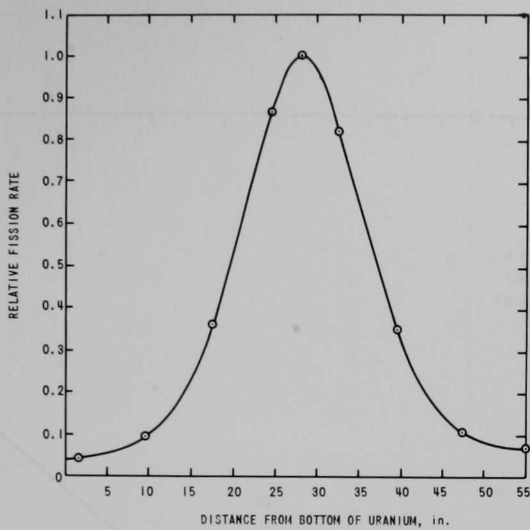


Fig. 8. Relative Axial Fission Rates in Depleted-uranium Element, Position 9A4

Figure 9 is a plot of % $\Delta V/V$ vs burnup. Considerable scatter is exhibited in the data, but as a first approximation, a straight-line relationship appears to exist between the values obtained.

The ratio of % $\Delta V/V$ to atom-percent burnup has been used by previous experimenters for describing the irradiation behavior of uranium. This ratio has been called the R value. In this study, R values were calculated by taking the average volume increase for each slug or section of slug and using the burnup value obtained for the midpoint of the slug section. Figure 10 plots the R values obtained against the axial position on the blanket element. Again, considerable scatter was present in the data, but the curve shown in Fig. 10 is the apparent "best fit" for the data obtained.

C. Metallographic and Electron-microscopy Results

The characteristic curves obtained by plotting the swelling of unalloyed uranium against burnup show a point where the swelling rate becomes asymptotic with small increases in burnup. Swelling above this point has been defined as "breakaway" or cavitational swelling, and the microstructure of the material above this point is characterized by grain-boundary tearing.

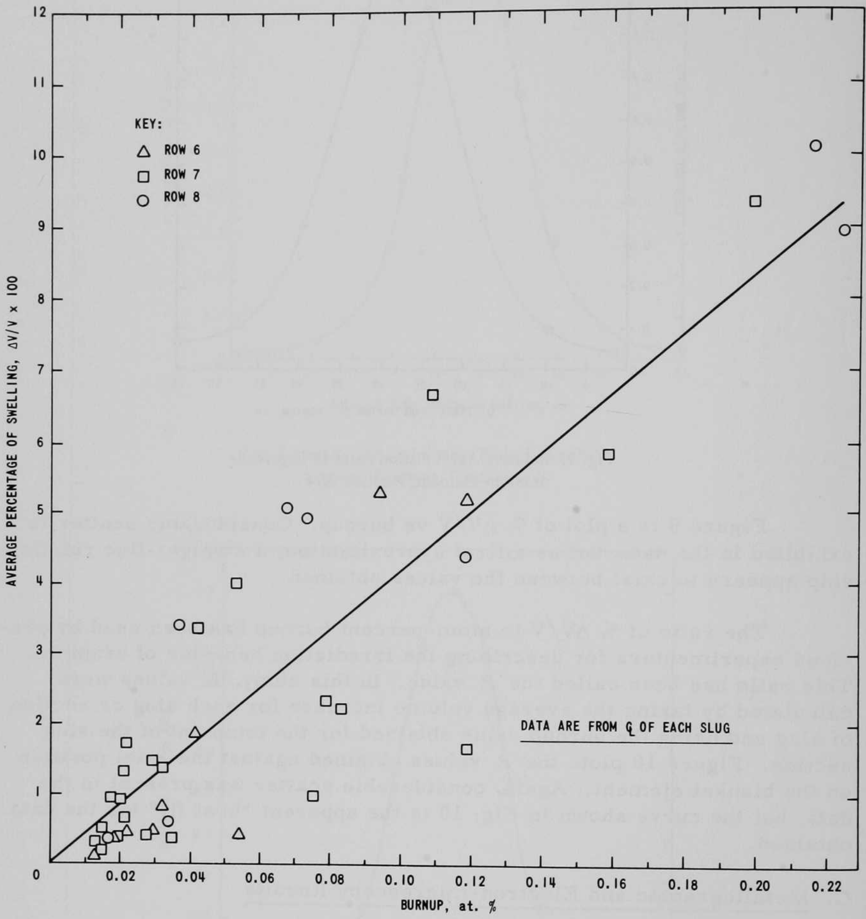


Fig. 9. Swelling vs Burnup for Selected Blanket Elements

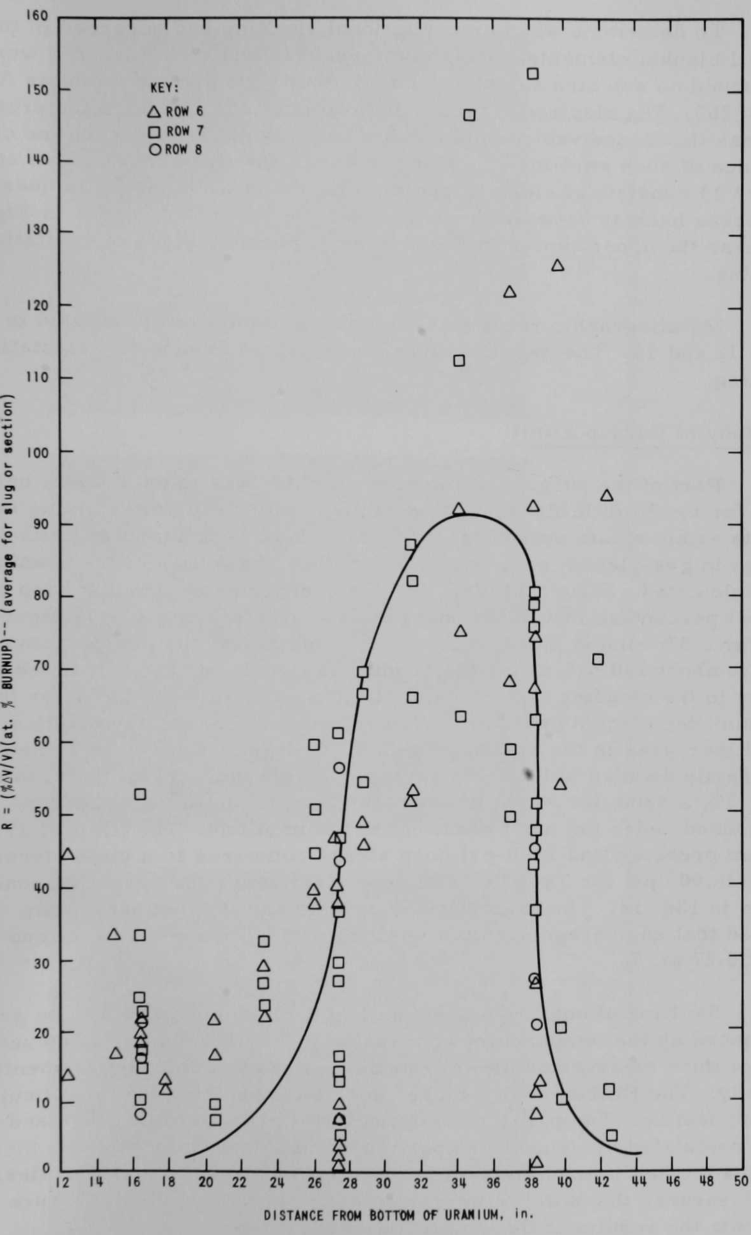


Fig. 10. R Values Obtained from Measurements of Density of Depleted-uranium Slugs

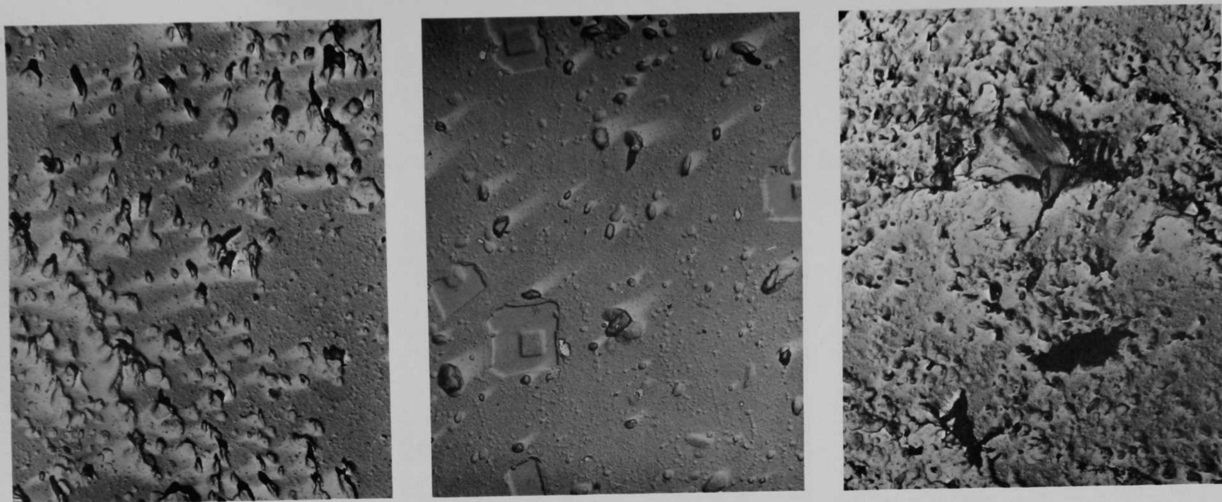
To determine whether cavitational swelling had occurred in the EBR-II blanket elements, electron microscopy and metallography were performed on selected samples from elements from subassemblies A-701 and A-760. The electron-microscopy examinations indicated the presence of areas that conceivably could contain cavitational swelling, but no clear evidence of such swelling was found in any of the five samples (see appendix). Figure 11 consists of electron photomicrographs of the areas in question. The areas halfway between the center and the lower left corner in Fig. 11a and near the upper center in Fig. 11c show possible signs of cavitational swelling.

Metallographic results on the same samples are presented in Figs. 12 and 13. The metallographs contained no evidence of cavitational swelling.

D. Study of Burnup Limit

Part of the purpose of the present study was to establish a burnup limit for the EBR-II blanket. To establish criteria for maintaining the burnup within a safe operating limit when cladding failure was unlikely, the change in gas-plenum pressure as a function of swelling of the uranium slugs was calculated. Figure 14 plots plenum pressure and cladding hoop stress against percentage of volume change of the uranium slugs. The figure shows that for a 3% volume increase in the uranium slugs, the gas-plenum pressure is about 150 psi, under the conditions listed, and the calculated hoop stress in the cladding is 2250 psi. Above a value of 3.5% $\Delta V/V$ for the uranium, the plenum pressure and hoop stress increase asymptotically with small increases in the uranium volume. Consequently, it was somewhat arbitrarily decided to limit the average volume increase of the uranium to about 3%, a value for which it was believed safe operating conditions would be assured under the most pessimistic assumptions. The limit of 150-psi plenum pressure and 2250-psi hoop stress compares to a yield stress of about 25,000 psi for Type 304 stainless steel under the operating conditions shown in Fig. 14. The examinations of elements from subassembly A-701 showed that an average uranium swelling of 3% occurred at a burnup of about 0.27 at. %.

Swelling of unalloyed uranium has been demonstrated to be greatly influenced by the temperature of irradiation.[4,5] Unfortunately, no actual temperature measurements were made on any of the blanket elements directly. The EBR-II design makes such measurements very difficult, if not impossible. Temperature distributions in the uranium slugs and cladding were calculated by using a computer code and data obtained from thermocouples located immediately above the tops of blanket subassemblies, where they measured the outlet temperature of the sodium coolant. Figure 15 presents the results of these calculations.[6]

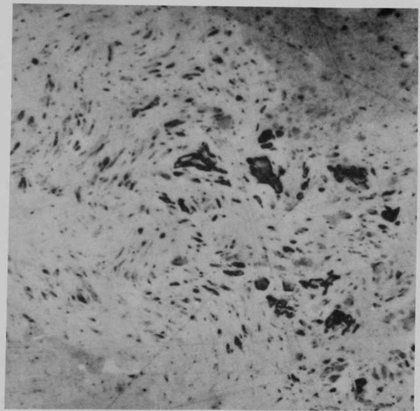


30-sec etch
(a) Specimen S-138

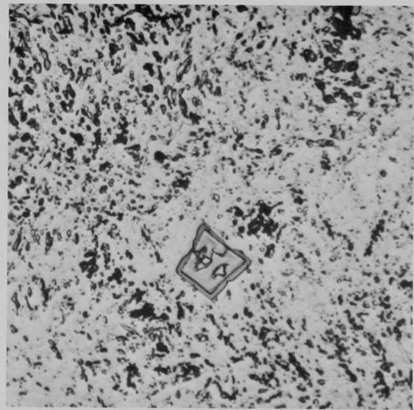
30-sec etch
(b) Specimen S-139

48-sec etch
(c) Specimen S-140

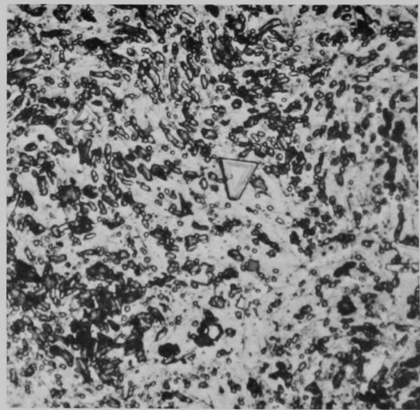
Fig. 11. Electron Photomicrographs of Irradiated Depleted-uranium Samples, Made by Replication Technique (all 2800X)



As polished

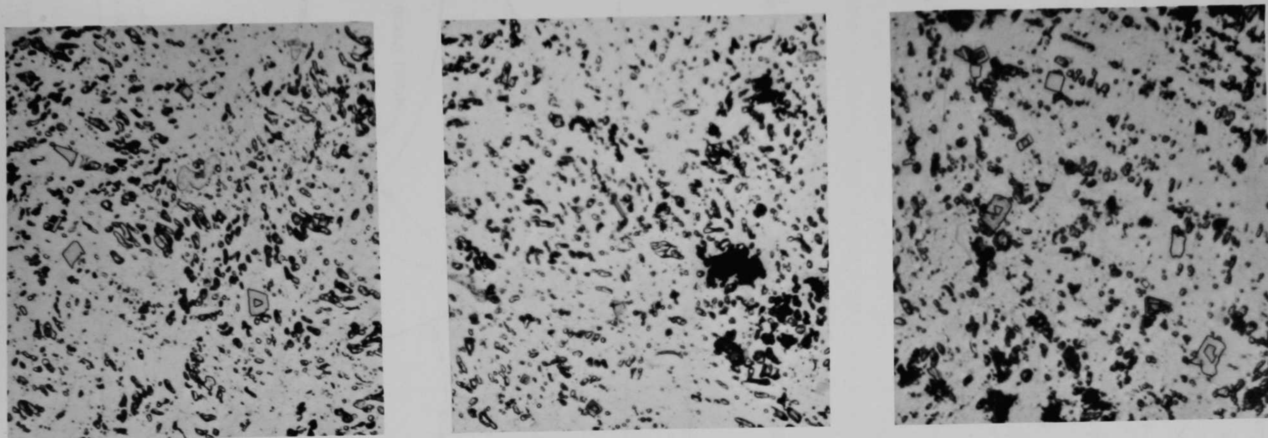


33-sec etch



48-sec etch

Fig. 12. Metallographs of Depleted-uranium Sample S-140 after 0.20 at. % Burnup (all 750X)



30-sec etch

60-sec etch

75-sec etch

Fig. 13. Metallographs of Depleted-uranium Sample S-138 after 0.26 at. % Burnup (all 750X)

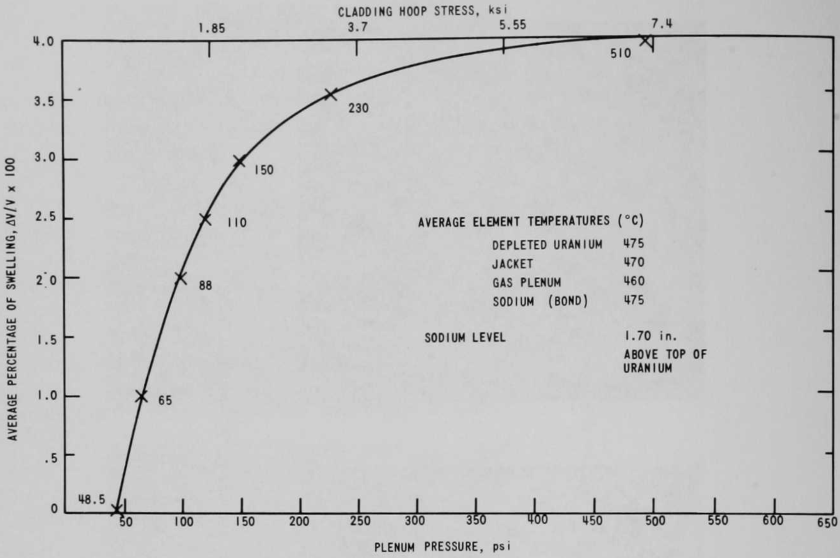


Fig. 14. Pressure vs Swelling for Radial-blanket Elements

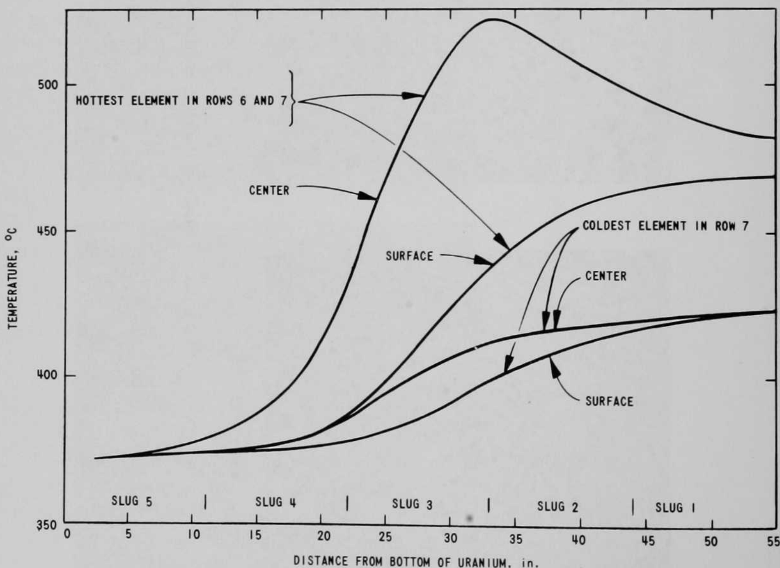


Fig. 15. Calculated Axial Temperature Distribution in Selected Blanket Elements[6]

VI. DISCUSSION OF RESULTS

Analysis of the data presented in this report indicates that the uranium slugs swelled isotropically. The uranium could swell unrestrained until it contacted the cladding or until the pressure was increased in the plenum region to values where swelling rates were affected.[6]

The results are limited by a lack of preirradiation data on the elements and uranium slugs. This lack of data increases the uncertainties of the shape and height of the curves of Figs. 9 and 10. However, the general shape of the curves should be as indicated, and these curves are in general agreement with the data in Ref. 5.

Present burnup limits appear to have been conservatively established. The 3% limitation on increase in volume of the uranium slugs results in a maximum hoop stress of about 2250 psi. At 950°F, a conservative value of the unirradiated yield stress for Type 304 stainless steel is about 25,000 psi. Therefore, the calculated hoop stress is less than one-tenth of the yield stress.

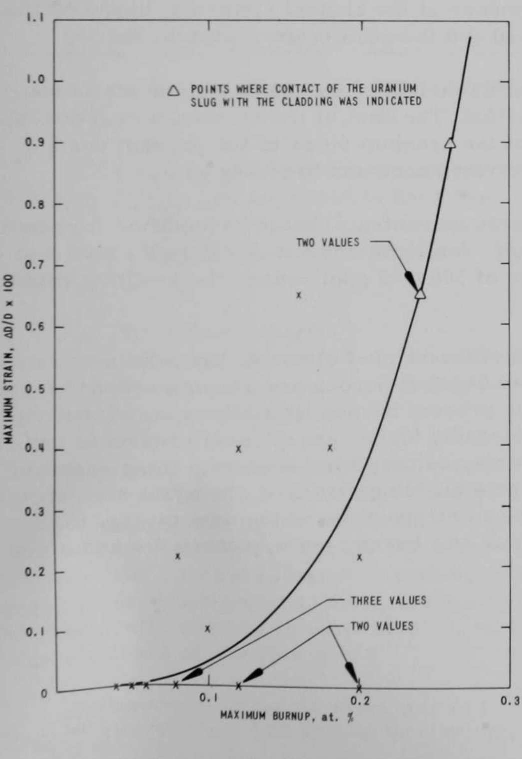


Fig. 16. Estimated Cladding Strain vs Maximum Burnup for Selected Blanket Elements

The 0.9% strain indicated in the cladding did not take into account such effects as swelling of the cladding. A cladding strain of about 2% may be attainable without greatly increasing the risk of cladding failure. As shown in Fig. 16, a wide variance in cladding strain for the same burnup levels was found. Some of this wide variance in cladding strain at similar burnup levels undoubtedly can be attributed to the lack of precise preirradiation data.

Cladding strain was considered to have occurred when (a) the diameter of the uranium slug was equal to or greater than the maximum specified ID of the as-fabricated cladding, i.e., when there were positive indications that the uranium was in contact with the cladding, and (b) the cladding OD in the area of uranium contact was larger than the OD in the areas where no uranium-cladding contact was apparent. There are only three data points in Fig. 16 where the uranium had definitely contacted the cladding. This indicates that the other strain values shown are due to swelling of the cladding or the lack of good preirradiation data.

VII. CONCLUSIONS AND RECOMMENDATIONS

To characterize more completely the swelling behavior of the depleted, unalloyed uranium-blanket elements in EBR-II, more examinations and experiments are necessary. The effects of temperature, pressure, and alloying constituents on swelling behavior should be studied under more carefully controlled conditions and using better preirradiation data.

The following conclusions are reached regarding the steps to be taken in optimizing the performance of the blanket elements, based on the examinations already performed and the results presented in Ref. 5.

1. Only about 14 in. of the height of the blanket elements is subjected to a significant neutron flux. The cost of the blanket elements could be reduced by using only two of the uranium slugs of the present design, without unduly compromising performance and breeding gain.

2. The silicon and aluminum contents should be modified to reduce the swelling rate of the uranium. A silicon content of 250 (±50) ppm and an aluminum content in a range of 500-700 ppm reduce the swelling rate of unalloyed uranium.[6]

With the present design of the blanket elements already in the reactor, incremental increases of 0.03-0.04 at. % burnup are recommended to evaluate the possibility of increasing the present burnup limitation. A new burnup limitation that will allow considerably higher gas-plenum pressures and a larger value of cladding strain may be possible. A burnup limit resulting in a hoop stress of 25,000 psi or a cladding strain of 2%, whichever occurs first, is recommended. Based on extrapolation of the data in Fig. 16, a cladding strain of 2% would occur at a burnup level between 0.3 and 0.4 at. %.

APPENDIX

Memorandum Concerning Metallographic Examination of Depleted-uranium Samples

24 January 1968

TO: V. G. Eschen — Metallurgy

FROM: W. F. Murphy — Idaho

SUBJECT: Metallographic Examination of Depleted-uranium Samples

Two irradiated depleted-uranium samples, S-138 and S-140, were sent by you to the attention of J. H. Kittel. You expressed an interest in the metallographic structure and evidence of cavitational swelling. The two samples were identified as follows:

Sample No.	Subassembly	Rod	Slug	Sample	Burnup, a/o
S-138	A-701	19	Middle	Top 1/2 in.	0.26
S-140	A-760	17	Middle	Top 1/2 in.	0.20

As received by us, the specimens were embedded in $1\frac{1}{4}$-in.-diameter bakelite mounts and appeared to have been polished and etched.

The two samples were repolished in the hot cells in Building 301 as follows:

9μ diameter paste, 3μ diamond paste, 1μ diamond paste, $1/4\mu$ diamond paste — washed with CCl_4 and dried with nitrogen gas between polishes.

The specimens were electrolytically etched in a solution consisting of 8 parts by volume of H_3PO_4, 5 parts ethylene glycol, and 5 parts ethyl alcohol. The time of etching was a variable, i.e., examination after 30-sec etch might be followed directly by another 30-sec etch and the total etching time would be 60 sec. The current involved in etching was not measurable with the equipment used. After etching, the specimens were rinsed in distilled water, ultrasonically cleaned in ethyl alcohol, rinsed in clean alcohol, and dried with a stream of nitrogen gas.

The metallographic structure of S-138 at 750X is illustrated in Fig. 1* after 30, 60, and 75 sec total etching time. For both of the longer

*The figures mentioned in this memorandum are not included in this report.

etching times, definite evidence of pitting is present. There are indications of the beginning of the swirled or worked structure typically seen in more highly irradiated uranium.

Figure 2 illustrates the structure of S-140 as polished and after etching for 33 and 48 sec total time. The appearance is similar to that of S-138. There are at least two phases present--the polyhedral particles and the matrix material. It is not clear that the globular clusters represent another phase. There is no clear evidence of cavitation swelling in their microstructures.

Primary replicas were made of the etched surfaces. These were shadowed and secondary replicas prepared. Electron photomicrographs of specific areas are shown in Fig. 3. There are no clear indications of cavitational swelling in any of the photographs. Figure 3(a) and (c) each contain areas which conceivably could be evidence of cavitation [(a), half way between center and lower left corner, and (c), upper center]. Figure 3(b) shows lines of small pits outlining specific areas. These could be the result of fission-gas bubbles. Examination of the replicas at higher magnifications up to 11,000X failed to reveal additional details.

REFERENCES

1. Koch, L. J., Loewenstein, W. B., and Monson, H. O., *Addendum to Hazard Summary Report--Experimental Breeder Reactor-II (EBR-II)*, ANL-5719 (Addendum) (June 1962).

2. Burt, W. R., Jr., McGowan, R. D., and Bean, C. H., *The Fabrication and Testing of Depleted Uranium Blanket Slugs for EBR-II*, ANL-6417 (Nov 1965).

3. Salley, R. L., and Bean, C. H., "The Assembly, Sodium Bonding, and Inspection of EBR-II Blanket Rods," *Transactions of the American Nuclear Society San Francisco 1960 Winter Meeting* *3*(2), 448-449 (Dec 1960).

4. Granata, S., and Saraceno, F., *The Relationship between Burnup, Temperature, and Swelling in Alpha Uranium*, J. Nucl. Mater. *9*(3), 367-368 (1963).

5. R. D. Leggett, Sect. 6.1, "Basic Swelling Studies," in *Reactor Fuels and Materials Development Programs for Fuels and Materials Branch of USAEC Division of Reactor Development and Technology*, Quarterly Progress Report, January-March 1966, BNWL-CC-694 (May 1966).

6. Cushman, R. A., personal communication (Feb 1967).